NRC & PSEG Meeting

Hope Creek Special Inspection and Technical Issues

January 12, 2005

This package includes the following documents:

1. Press Release I-05-02 (ADAMS Accession No. ML050100266)
2. Hope Creek Nuclear Generating Station - Preliminary Results of the Special Inspection for the October 10, 2004 Event (ADAMS Accession No. ML050100182)
3. Hope Creek Nuclear Generating Station - Summary of Results of NRC Review of Technical Issues (ADAMS Accession No. ML050100194)
4. Hope Creek, PSEG Actions in Response to NRC Concerns Regarding 'B' Reactor Recirculation Pump (ADAMS Accession No. ML050100288)
5. Confirmatory Action Letter 1-05-001 Hope Creek Generating Station (ADAMS Accession No. ML050120012)
6. Meeting Feedback Form

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

Office of Public Affairs, Region I
475 Allendale Road, King of Prussia, Pa.
Web Site: http://www.nrc.gov/OPA

No. I-05-002 January 10, 2005

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NRC RELEASES TWO LETTERS REGARDING HOPE CREEK NUCLEAR PLANT; LETTERS DISCUSS PUMP REVIEW AND SPECIAL INSPECTION RESULTS

The Nuclear Regulatory Commission has released two letters regarding the Hope Creek nuclear power plant in advance of a public meeting about the facility scheduled for this Wednesday, Jan. 12. One letter provides the agency's assessment of whether one of Hope Creek's two reactor recirculation pumps can safely operate until the plant's next refueling and maintenance outage. The other letter provides the preliminary results of an NRC special inspection conducted at the Hancocks Bridge (Salem County), N.J., plant following a steam pipe failure and shutdown with complications that occurred there on Oct. 10.

With regard to the NRC's assessment of the plant's "B" recirculation pump, the NRC has conducted a detailed review of its safety. This review included analyses of information provided by PSEG, which operates the plant, meetings with the company to obtain additional data and assessments of the information by the agency's technical experts.

Following this review, the NRC has concluded that operation of the pump for one more operating cycle is acceptable, provided that PSEG implements commitments to closely monitor the pump for vibrations throughout the cycle and to respond promptly to any evidence that its performance may be degrading. (Nuclear plants typically shut down for refueling and maintenance outages about once every 18 to 24 months.) These commitments will be formalized in a Confirmatory Action Letter that the NRC expects to issue to PSEG prior to the plant's restart. The NRC's Resident Inspectors at the site also will ensure that PSEG complies with this new commitment.

"After a careful and thorough review, including analysis by the NRC's Office of Nuclear Reactor Regulation, the NRC concludes that the Hope Creek nuclear plant can operate safely without replacing the pump until its next refueling and maintenance outage," NRC Region I Administrator Samuel J. Collins said. "That conclusion, however, is contingent on requirements agreed to by the company that rigorous and continuous monitoring be maintained of pump parameters, including vibration levels, so that prompt actions can be taken should there be abnormal indications."

Following the Oct. 10th event at the plant, an NRC team of five full-time and four part-time members was tasked with evaluating the circumstances surrounding it. The review included, among other things, an assessment of whether the failure of the 8-inch-diameter pipe could have been prevented and an independent evaluation of equipment and human performance issues that complicated the shutdown of the reactor.

Based on the preliminary results of that inspection, the team has determined that Hope Creek operators successfully responded to the event and that PSEG conducted comprehensive follow-up evaluations and developed appropriate corrective actions. The NRC team also has confirmed PSEG's determination that the root cause of the event was that plant personnel did not properly evaluate and address a degraded level control valve for the moisture separator drain tank. That valve malfunctioned several weeks prior to the event when it opened and then failed to close. As a result, a drain line for the tank received a combination of water and steam even though it is primarily intended for the removal of water. The ensuing turbulent flow placed stresses on the pipe that eventually led to its failure and caused the plant's Oct. 10th shutdown.

The NRC is continuing to review the significance of this finding and will issue a preliminary determination in the full inspection report to be issued within 45 days after the Jan. 12th meeting. However, the team believes the finding could be of low to moderate safety significance. If validated through the agency's review process, the finding would eventually be finalized as a "white" issue and lead to some additional NRC oversight. (The NRC's Reactor Oversight Process classifies inspection findings by color, ranging from "green," for a very low safety issue, to "red," for a high safety issue.)

In addition, the team has identified three other inspection findings associated with some equipment issues that challenged plant operators during the shutdown. The team has deemed those issues to be of very low safety significance because none of the problems would have prevented the systems involved from performing their intended safety functions.

Copies of both reports will be posted on the NRC's web site at: http://www.nrc.gov/reactors/plant-specific-items/hope-creek-salem-issues.html .

As previously announced, the Jan. 12th meeting, which will be open to the public for observation, is scheduled to begin at 7 p.m. at the Holiday Inn Select Bridgeport, located off Exit 10 of Interstate 295 in Swedesboro, N.J. Before the session is adjourned, NRC staff will accept questions and comments from the public.

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January 10, 2005

Mr. A. Christopher Bakken, III
President and Chief Nuclear Officer
PSEG Nuclear LLC - N09
P. O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK NUCLEAR GENERATING STATION - PRELIMINARY RESULTS OF THE SPECIAL INSPECTION FOR THE OCTOBER 10, 2004 EVENT

Dear Mr. Bakken:

During the period of October 14 through December 16, 2004, the U.S. Nuclear Regulatory Commission (NRC) conducted a special inspection at the Hope Creek Nuclear Generating Station in accordance with Inspection Procedure 93812, "Special Inspection." This special inspection was conducted to assess the circumstances surrounding an event that occurred on October 10, 2004. Specifically, the plant was manually shutdown due to the failure of an 8-inch diameter moisture separator drain line, which discharges to the main condenser. By letter dated October 17, 2004, you provided the NRC with an overview of your plans to respond to this event. The NRC acknowledged your correspondence by letter dated October 21, 2004, from Samuel J. Collins, Region I Administrator. In that letter, the NRC stated that due to the heightened stakeholder interest in the event and consistent with NRC's openness strategic goal, the NRC would publish the preliminary results of the special inspection and meet with the public to review your actions and NRC findings prior to start-up of the Hope Creek facility.

The enclosure to this letter provides a summary of the inspection scope and preliminary inspection results in the areas reviewed. Please note that the final inspection results, including the number of findings and characterization of their significance, may change based on additional information and further review. The final inspection results, including any associated regulatory compliance issues, will be documented in NRC Inspection Report 05000354/2004013 which will be issued within 45 days after the inspection exit meeting scheduled for January 12, 2005.

The inspection focused on Hope Creek's investigation and root cause evaluations, including issue identification, extent of condition, potential common cause failures, root causes and corrective actions. The team independently evaluated the equipment and human performance issues that complicated the response to the event and assessed compliance with technical specifications and the emergency plan. Team members also evaluated the radiological releases associated with the event.

The team determined that PSEG's root cause evaluations were comprehensive and appropriately considered potential causes, extent of condition, and the problems encountered during the event. The inspection team confirmed that the root cause of the event was that personnel did not properly evaluate and address a degraded level control valve for the moisture separator drain tank. The level control valve malfunctioned several weeks prior to the event and caused the moisture separator drain system to operate in a condition outside its design. As a result, an 8-inch pipe in that system failed and caused the event on October 10, 2004. The assessment of this finding remains under review, but preliminarily, the finding is of low to moderate safety significance because it resulted in an actual plant event that included the loss of the normal power conversion system (the main condenser).

Overall, the team found that operator response to the transient was acceptable; however the operators were challenged by some equipment issues during the response to the event. Although these equipment problems challenged the operators, none of the problems would have prevented the systems from performing their intended safety functions or rendered the systems inoperable. The NRC inspection team identified three findings of very low safety significance associated with equipment and operational issues.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its Enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS).

Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Enclosure: Summary of Inspection Scope and Preliminary Results

Docket No. 50-354
License No. NPF-57

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Inspection Scope and Preliminary Results

A. Inspection Scope

During the period from October 14 through December 16, 2004, the NRC conducted a special team inspection in accordance with NRC Inspection Procedure 93812, "Special Inspection," at the Hope Creek Nuclear Generating Station. The special inspection was conducted to assess the circumstances surrounding an event that occurred on October 10, 2004, involving the failure of an 8-inch moisture separator drain line, which discharges to the main condenser. The special inspection was initiated in accordance with NRC Management Directive 8.3, "NRC Incident Investigation Program," based on deterministic criteria and an initial risk assessment. Specifically, the condition involved possible generic implications, and involved questions pertaining to licensee operational performance. The initial risk assessment for this event was in the range where a special inspection was warranted.

The special inspection team consisted of five full-time members with expertise in the areas of plant operations, materials and mechanical engineering, and corrective actions. There were also four part-time members with expertise in the areas of emergency preparedness, radiological controls and protection, materials engineering, and probabilistic risk assessment.

The special inspection team was tasked to evaluate PSEG's analysis of the cause of the moisture drain pipe failure, extent of condition and actions to prevent recurrence, as well as to determine whether prior opportunities were available to prevent the event. The team was also tasked to develop an event chronology and independently evaluate human and equipment performance issues that complicated the response to the event. The inspectors also reviewed compliance with procedures and verified radiological releases were within regulatory requirements.

B. Preliminary Inspection Results

1. Plant Response: Personnel and Equipment Performance

The team reviewed and assessed licensed operator performance during the transient initiated by the moisture separator drain line failure until the plant was placed in the cold shutdown condition. The team provided particular focus on equipment issues that challenged the operators during the event. The team performed a detailed review of the data related to the event to assess overall equipment and human performance.

Results:

The team found that, overall, the operator response to the event was acceptable. However, there were some equipment issues that challenged the operators during the event and associated recovery.

Three findings that have been preliminarily determined to be of very low safety significance and one minor operator performance issue are described below.

<u>High Pressure Coolant Injection (HPCI) System Valve Malfunction</u>

PSEG determined that a limit switch had been incorrectly set for one of the two closed valves needed to satisfy an interlock to allow the HPCI full flow test valve (F008) to open. This self-revealing problem caused a delay of about five minutes when the operators attempted to place the HPCI system in service to control reactor pressure. The operators were able to satisfy the interlock and open the F008 valve by sending an additional close signal to the closed valve. The finding was considered to be of very low significance because it did not impact the accident mode of operation for the HPCI system, reactor pressure remained relatively stable during the period of time when HPCI operation was delayed, and alternate pressure control methods were available.

<u>Reactor Core Isolation Cooling (RCIC) System Flow Oscillations</u>

PSEG determined that operating experience regarding low flow limitations while operating the RCIC system in automatic flow control had not been incorporated into system operating procedures and operator training. As a result, the RCIC system was operated in a low flow condition (about 350 gpm) while in the automatic flow control mode and experienced unexpected oscillations. The RCIC system is normally aligned to operate in the automatic flow control mode in a high flow condition. During the event, the RCIC system had to be secured for approximately 10 minutes until the control system was adjusted. This problem was determined to be of very low significance since the RCIC system remained capable of performing its required safety functions, reactor water level was always maintained at least ten feet above the top of the active fuel, and the HPCI system was available for reactor vessel level makeup.

<u>HPCI System Vacuum Pump Trip</u>

PSEG determined that the wrong lubricant had been applied to the HPCI vacuum pump shaft. As a result, the HPCI system barometric condenser vacuum pump tripped several times during the depressurization and cooldown phase of the event. While the vacuum pump problem did not render the HPCI system incapable of performing its safety function, the operators decided to remove the HPCI system from service to prevent the release of radioactive effluents into the HPCI room due to operation without the vacuum pump. The finding was determined to be of very low significance because the HPCI system remained operable to perform its safety function without the vacuum pump.

Technical Specification Action Statement Interpretation

The team identified that the operators misinterpreted the Technical Specification Action Statement time requirement to place the plant into a cold shutdown condition within 24 hours of declaring the residual heat removal (RHR) system inoperable. During the event the operators aligned the RHR system to cool the suppression pool and declared the system inoperable in accordance with plant operating procedures. However, the operators misinterpreted the Technical Specification Action Statement time requirement and believed that they had 36 hours to complete the plant cooldown. The team believes this issue was of minor significance since the plant cooldown was completed safely and the RHR system could have been realigned to provide reactor makeup if needed.

2. Moisture Separator Drain Tank Piping Failure

The team reviewed the moisture separator drain tank system to determine how its design and operation may have contributed to this event. The team also reviewed the history of design and operational challenges associated with the system to determine if there were prior opportunities to identify, evaluate and prevent the conditions that led to the event.

Results:

The team found that engineers did not properly evaluate and recommend appropriate actions for a moisture separator drain tank level control valve problem. The level control valve problem resulted in a degraded condition that was outside of the system design basis and led to the steam pipe failure. The significance of this finding remains under review in accordance with the NRC's process for evaluating the significance of inspection findings.

Inadequate Evaluation and Corrective Action for Degraded Condition

PSEG determined that engineers did not identify that operation of the plant with the 'A' moisture separator drain tank level control valve (LV-1039A) failed open was outside the system design basis. Specifically, valve LV-1039A failed open on September 16, 2004, however, engineers did not recognize that continued operation in this condition placed the moisture separator drain system in a condition beyond its design capability. The open valve allowed the moisture separator drain tank to drain down which resulted in two-phase flow (a mixture of steam and water) through the moisture separator drain line to the main condenser. The two-phase flow introduced dynamic loading effects that had not been considered as part of the original design basis. This high dynamic loading caused the 8-inch moisture separator dump line to fail on October 10, 2004. In addition, engineers did not recognize, evaluate and properly address the fact that a similar condition occurred in 1988 and led to a crack in the same line.

The final disposition of this finding remains under review, however, preliminarily, it appears to be of low to moderate safety significance because the failure to correct the degraded condition resulted in an actual plant event that included the loss of the normal power conversion system (the main condenser). The main condenser was manually isolated by operators to terminate the steam break and required operators to use alternate means to depressurize and cooldown the plant.

3. Root Cause and Corrective Actions

The team evaluated PSEG's formal root cause evaluations associated with this event, including efforts to identify the cause of the pipe rupture, extent of condition reviews, and actions to prevent recurrence. The team independently evaluated personnel actions and equipment performance to assess the adequacy of PSEG's investigation.

Results:

The team determined that PSEG's root cause evaluations were comprehensive and appropriately considered potential causes and extent of condition for the pipe failure and the problems encountered during the event. The team determined that PSEG's proposed corrective actions were appropriate to address the identified problems and confirmed that corrective actions necessary for restart were implemented. The corrective actions included, in-part: a revised engineering decision making process, field walkdowns and inspections of pipe hangers and components, and revised operating and maintenance instructions to address the equipment problems which challenged operators during the event. There were no findings identified in this area.

4. Radiological Assessment

The team reviewed data and calculations used to quantify the amount of radioactive material released as a result of this event.

Results:

There was a small radiation release from the plant as a result of this event that was well below federally approved operating limits. Specifically, the total radiological release rate was less than 2% of Technical Specification limits. The total amount released was approximately 9.2 Curies of noble gas and consisted of both monitored and unmonitored release paths. A typical release for the same time period during normal operation would have been about 4.9 Curies. The unmonitored release occurred during a relatively short time frame (approximately 50 minutes) when steam was released to and exited the turbine building without transiting through the monitored ventilation exhaust path.

The team concluded that the radiological consequences of this event were negligible, and there were no findings identified in this area.

January 10, 2005

Mr. A. Christopher Bakken, III
President and Chief Nuclear Officer
PSEG Nuclear LLC - N09
P. O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK NUCLEAR GENERATING STATION - SUMMARY OF RESULTS OF NRC REVIEW OF TECHNICAL ISSUES

Dear Mr. Bakken:

The purpose of this letter is to communicate the results of the Nuclear Regulatory Commission (NRC) staff review of two technical issues at the Hope Creek Nuclear Generating Station. Specifically, the first issue was associated with the 'B' reactor recirculation pump which has historically exhibited elevated vibration levels and has experienced other problems such as premature seal degradation. The second review looked at whether the exhaust piping for the high pressure coolant injection system had experienced a significant transient (i.e. a water hammer) that could have damaged the piping during the last refueling outage.

The enclosures to this letter provide summaries of the staff review of these technical issues. For the recirculation pump vibration issue, the staff concluded that your proposed continuous monitoring program for the Hope Creek reactor recirculation pumps provided reasonable assurance that a potential crack could be detected in time to allow the operators to take appropriate corrective actions prior to a shaft failure. We intend to confirm commitments related to the monitoring of the "B" recirculation pump and replacement of the pump shaft in a confirmatory action letter which will be issued prior to the re-start of Hope Creek.

For the high pressure coolant injection system exhaust line issue, the staff concluded that there was reasonable assurance that the high pressure coolant injection system exhaust line integrity had not been challenged by a "water hammer" event. Final documentation of these results will be included in NRC Inspection Report 05000354/2004005 which we expect to issue within 45 days.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its Enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS).

Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Enclosures:
1. Summary of Recirculation Pump Vibration Review
2. Summary of High Pressure Coolant Injection Exhaust Line Review

Docket No. 50-354
License No. NPF-57

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Enclosure 1

Reactor Recirculation Pump Vibration Review

Background

The "B" Hope Creek reactor recirculation (RR) pump has had a historical problem involving high vibration levels—about double those on the "A" RR pump. Past licensee actions to reduce the vibration levels have not been effective. The high vibrations have been attributed, in part, to a slight bowing of the shaft in the area below the seal package area. The vibrations have led to frequent seal replacements (1.5-year intervals versus the expected 6-year intervals).

In addition to the bowing, the "A" and "B" RR pump shafts are expected to have some degree of thermally induced stress cracking based on industry operating experience described in GE Service Information Letter (SIL) 459. GE SIL 459 recommends three actions to address this problem: vibration monitoring, shaft inspections after about 80,000 hours of operation and action to mitigate the thermal stress initiators. Hope Creek's RR pumps have over 130,000 hours of operation, and PSEG has not performed the recommended inspections.

In addition to the pump vibrations, there are vibrations on the associated RR and RHR system piping which have resulted in damage to system sub-components (MOV handwheel and limit switches). To date none of the vibration-induced component problems have rendered any safety-related system inoperable.

Sargent and Lundy (S&L) performed an independent assessment for PSEG which concluded that return of Hope Creek to service for the next operating cycle was acceptable given the current level of RR pump and piping vibrations. S&L's conclusion was based upon data which indicated that the vibration level for Hope Creek's "B" RR pump was consistent with RR pumps at other facilities and also based on an assumption that operators would be able to respond to an increasing vibration trend and take action to remove the pump from service prior to shaft failure.

The S&L assessment is summarized in the report, "Independent Assessment of Hope Creek Reactor Recirculation System and Pump Vibration Issues," dated November 12, 2004. The staff reviewed the S&L report and developed a number of questions which were provided to the licensee on December 1, 2004. PSEG responded to the questions during a December 17, 2004, public meeting with the NRC. PSEG provided an additional response to the staff questions in a December 22, 2004, submittal. In addition, numerous teleconferences were held between PSEG and the NRC in December 2004 and January 2005 to discuss the "B" RR pump vibration issue.

The S&L Report concluded that there is no immediate need to replace the "B" pump rotor during the current refueling outage. S&L recommended that both pumps be monitored for vibrations and that a rapid rise in vibrations would be a sufficient reason to shut the pump down immediately for an internal inspection and shaft replacement, as the window between the rise in vibration and potential shaft failure is expected to be small.

PSEG also provided additional background information in Report H-1-BB-MEE-1878, "Hope Creek 'B' Recirculation Pump Vibration Analysis," Revision 1, dated December 16, 2004. The report concluded that, while the "B" RR pump has elevated vibrations when compared to the industry average, these vibration levels are not detrimental to the operation or reliability of the pump. The report also indicated that, although the risk of a RR pump shaft cracking event during any given cycle cannot be quantified, the operating experience of 29 RR pumps in operation longer than the Hope Creek "B" RR pump provides sufficient data to conclude that the risk of a shaft cracking event during the next cycle is minimal.

Staff Review

The staff review focused on the following key issues regarding the RR pump operation:

(1) Does PSEG have a technical evaluation which shows that the RR pumps can be operated for another cycle without failure of the shafts considering the identification of shaft cracks that have been observed at other facilities with the same design RR pumps?

(2) Can PSEG provide data which demonstrates that shaft cracks have been detected at other facilities with the same design RR pumps using vibration monitoring? Can the cracks be detected in time for the operators to take appropriate actions?

(3) What are the consequences of a RR pump failure during plant operations?

GE SIL 459 indicates that all Byron Jackson RR pump shafts inspected have shown some degree of thermally induced cracking. The cracking occurs near the pump thermal barrier where mixing of cold seal purge system water and the hot reactor coolant water occur. The cracks initiate as axial cracks in the pump shaft. The licensee indicated that, if the cracks remain axial, the cracks will grow slowly and not affect the operation of the pump. However, the licensee also indicated that given sufficient mechanical loads, the cracks can become circumferential. The circumferential cracks can propagate to shaft failure under mechanical loading. The time it takes to transition from slow growing axial cracks to more rapidly growing circumferential cracks depends on the magnitude of the mechanical loads acting on the pump shaft. Since the licensee does not know the magnitude of the mechanical loads, it is difficult to predict the shaft life based on the magnitude of the operational loads.

The licensee has cited operating experience of other BWRs with similar Byron Jackson RR pumps. The licensee indicates that the age of the Hope Creek RR pumps is about average for the pumps of similar design at other BWRs. The staff notes that a number of the older pumps included in the licensee's comparison are much smaller than the Hope Creek pumps. While the operating experience provides some confidence that the pumps can be safely operated beyond the time interval recommended in GE SIL 459, the crack growth analyses provided by the licensee indicate that the time is highly dependent on the magnitude of the mechanical loads which are not well known.

The licensee also provided the level of vibration recorded at other BWRs with similar Byron Jackson RR pumps. The licensee concluded that measured vibration levels of the Hope Creek RR pumps are within the range of the vibration levels measured at other BWRs. However, the level of vibration of the "B" pump is toward the high end of the range of vibration levels measured at other BWRs. Therefore, the "B" pump is experiencing higher vibratory loadings than most of the pumps in the licensee's survey. In addition, the licensee cited a history of problems in its attempt to balance and align the pump shaft. These problems caused additional mechanical loadings on the pump shaft which could increase the potential for circumferential cracks to have developed in the shaft. On the basis of the above discussion, the staff concludes that the probability of a pump shaft failure of RR pump "B" during the next cycle of operation is indeterminate based on PSEG's evaluation of the potential thermal and mechanical loads on the pump shaft.

The licensee relies on vibration monitoring to detect circumferential cracking of the RR pump shaft with sufficient lead time for operators to secure the pump from complete shaft failure. The licensee developed a plan for monitoring the vibration levels of the RR pumps. The key elements of the plan involve continuous basic monitoring of the overall level of vibration and continuous monitoring of the vibration harmonics for enhanced detection capability of potential shaft cracking.

The licensee's continuous basic vibration level monitoring by the operations department consists of a pump vibration alarm and pump speed reduction if the "B" pump vibration level reaches 11 mils (0.011 inch), and removal from service if the pump vibration level reaches 16 mils (0.016 inch). The continuous monitoring of the vibration harmonics consists of pump vibration alarms and pump speed reduction if the synchronous speed (1X) vibration amplitude, two times synchronous speed (2X) vibration amplitude, 1X phase angle, or 2X phase angle exceed defined allowable limits. If the monitored values do not fall within their allowable limits at the reduced pump speed, the licensee will remove the RR pump from service. The allowable limits are established using ASME OM Standard, "Reactor Coolant and Recirculation Pump Condition Monitoring." The licensee will record baseline data to establish these allowable limits during plant startup. The licensee provided two technical papers in support of the proposed vibration monitoring criteria.

The first technical paper is entitled, "Case History Reactor Recirculation Pump Shaft Crack," Machinery Messages, December 1990. The paper discusses the RR pump shaft cracking experience at the Grand Gulf nuclear power plant. The paper indicates that the vibration level increased rapidly over a three hour period before the pump was secured at slow speed. Although the shaft did not experience a complete failure, subsequent inspection revealed the shaft was cracked approximately 320 degrees around the circumference. The paper indicates that it is necessary to monitor the 1X and 2X steady state vectors (1X and 2X amplitudes and phase angles) on a continuous basis and to compare these monitored values to an acceptance criteria. The paper also indicates that alarms are necessary to alert the user to amplitude and phase deviations that are outside the acceptance criteria.

The second paper is a Technical Bulletin from Bently, Nevada, "Early Shaft Crack Detection on Rotating Machinery Using Vibration Monitoring and Diagnostics." The technical bulletin indicates that shaft cracking can be detected by monitoring the 1X and 2X vectors. The technical bulletin also recommends continuous monitoring of machines that are susceptible to shaft cracking.

These papers recommend using continuous monitoring of the 1X and 2X vectors as a predictive method to detect significant shaft cracking. The staff requested that the licensee provide some evidence that vibration monitoring was effective for detecting shaft cracks in RR pumps similar to the Hope Creek RR pumps. The licensee cited the experience at Grand Gulf discussed above. The Grand Gulf RR pump shafts are hollow shafts as opposed to the solid shafts used in the Hope Creek RR pumps. Therefore, the Grand Gulf experience may not be directly applicable to Hope Creek. The licensee provided additional information which indicates that cracks in reactor coolant pump shafts were identified at Sequoyah (technical presentation to NDE Steering Committee by G. Wade, July 12, 2002) and Palo Verde Unit 1 (Palo Verde Nuclear Generating Station Cracked Reactor Coolant Pump Shaft Event, H. Maxwell, 1996) using vibration monitoring. Although these plants are Pressurized Water Reactors (PWRs), the reactor coolant pumps have solid shafts. The licensee indicated that these pumps had operated for a significant period of time after the first indication of shaft cracks by vibration monitoring. A staff review also identified that vibration monitoring successfully identified a reactor coolant pump shaft cracking at St. Lucie Unit 2 (LER Number: 1993-005). The PWR reactor coolant pump experience provides some indication that a solid pump shaft will provide better early crack detection capability than the hollow pump shafts, such as those used at Grand Gulf. PSEG has provided data which demonstrates that shaft cracks in pump shafts similar to those used at Hope Creek have been detected at other facilities, and that these cracks were detected in time for operators to take appropriate actions.

On the basis of the available operating experience, the staff concludes that continuous monitoring of the 1X and 2X amplitudes and phase angles provides reasonable assurance that circumferential shaft cracking can be detected with sufficient time for the plant operators to take appropriate actions. The licensee will either reduce the RR pump speed or remove the pump from service if the monitoring system detects vibration levels that exceed the limits specified in the vibration monitoring plan.

The staff also reviewed the licensee's assessment of the potential consequences of a RR pump shaft failure. The RR pump shaft axial cracking that has been reported occurred below the seal area and above the pump hydrostatic bearing. This is the region where a potential RR pump shaft failure would be expected to occur. The pump impeller would be expected to settle at the bottom of the pump casing, which could potentially result in some damage to the pump casing. The unsupported end of the upper part of a broken shaft may damage the shaft seal. A seal failure would result in leakage of reactor coolant through clearances around the upper half of the broken pump shaft. This leakage would be bounded by the design basis small LOCA event. If such an event were to occur, the licensee would be able to isolate the pump using the RR loop isolation valves, thereby terminating any reactor coolant system leakage.

Conclusion

The staff concludes that the licensee's continuous monitoring program for the Hope Creek RR pumps, as discussed above, provides reasonable assurance that a potential crack in the RR pump shaft can be detected in time for operators to take appropriate actions to reduce the pump speed or remove the RR pump from service prior to a complete shaft failure.

Enclosure 2

High Pressure Coolant Injection (HPCI) Exhaust Line Review

Background

On November 1, 2004, with the plant in Mode 5 for refueling outage 12, tandem snubbers from the HPCI turbine exhaust piping failed during dynamic testing. A followup inspection of the HPCI piping resulted in the observation of a damaged pipe support and a snubber anomaly that could have been the result of a water hammer event in the HPCI turbine exhaust line. A subsequent licensee evaluation of the reported observations found that there was no conclusive evidence that a water hammer had occurred in the HPCI turbine exhaust line.

Staff Review

The licensee provided an assessment of the tandem snubber failures performed by the snubber manufacturer, Lisega. The snubber failures occurred in the fluid reservoirs. Lisega indicated that the fluid reservoir failures were caused by stuck poppet valves that allowed fluid to leak into the reservoir during testing. Lisega concluded that repeated testing of the HPCI snubbers in compression resulted in over-pressurization of the reservoirs. Lisega also indicated that the snubbers would have functioned in response to a seismic event. The licensee's assessments of the other observations, identified during the initial inspection of the HPCI exhaust line, provided reasonable dispositions of the observed conditions.

A licensee inspection of the accessible portions of the HPCI exhaust line in the turbine room and the torus room found no evidence of large pipe distortion or excessive pipe movement at support locations which likely would have been present if a water hammer had occurred. This was confirmed by the NRC inspectors. The licensee also performed non-destructive examination (NDE) of all field welds on the 20 inch HPCI exhaust line. All welds were found satisfactory. The inspections and weld examinations performed by the licensee are the type of actions the NRC staff would require after a water hammer event.

Conclusion

The licensee provided plausible explanations for the snubber failures that occurred during snubber testing and for the identified support damage and snubber anomaly identified during the followup HPCI inspection. In addition, the licensee performed the type of inspections and NDE examinations that the NRC would require after a water hammer event and found no adverse results. Therefore, the staff concluded that there was reasonable assurance that the integrity of the HPCI exhaust line had not been challenged by a water hammer event.

LR-N05-0017
January 9, 2005

Mr. Samuel Collins, Regional Administrator
United States Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406-1415

PSEG ACTIONS IN RESPONSE TO NRC CONCERNS REGARDING 'B' REACTOR RECIRCULATION PUMP HOPE CREEK GENERATING STATION DOCKET NO. 50-354

Reference: Telecon Mr. Chris Bakken, PSEG, and Mr. Sam Collins, USNRC January 7, 2005

Dear Mr. Collins:

In response to our telephone discussions on January 7, 2005, this letter documents PSEG Nuclear LLC's (PSEG) commitments regarding the Hope Creek 'B' Reactor Recirculation Pump.

1) PSEG will implement a vibration-monitoring program to continuously monitor the 'B' Reactor Recirculation Pump's primary and secondary harmonic parameters (total amplitude, 1X and 2X amplitude, and 1X and 2X phase angle) during future operating cycles. This program shall include establishing objective criteria that demonstrate that monitored parameters are within acceptable range and developing procedures which specify the actions to be taken if the monitored parameters are outside of the specified range of the acceptance criteria. We understand that PSEG procedures HC.OP-AB.RPV-0003(Q), HC.OP-AR.ZZ-0008(Q) Attachment E-4, and HC.ER-AP.BB-0001(Z) Rev. 0, provided by PSEG letter dated January 4, 2005 meet this commitment.

   Further, this program will continue until an inspection of the 'B' Reactor Recirculation Pump's rotating assembly and replacement of the pump shaft have been completed.

2) PSEG will notify the NRC prior to implementing any change to the vibration monitoring and operating procedures cited above. This notification will provide sufficient time for the NRC to complete a review of the proposed changes.

3) PSEG will replace the 'B' Reactor Recirculation Pump shaft and inspect the pump's rotating assembly and pressure boundary components (such as the pump casing and cover) at the earlier of the next refueling outage (RFO13) or during an outage of sufficient duration to accomplish pump replacement.

During the current refueling outage (RF012) PSEG has completed extensive work directed toward improving equipment reliability and correcting long-standing problems. Attachment 1 provides a summary listing of some of the more significant activities.

Should you have any further questions please contact Christina Perino, Director-Licensing and Nuclear Safety at 856-339-1989.

Very truly yours,

A. Christopher Bakken, III
President and Chief
Nuclear Officer

Attachment

C Mr. D. Collins, Project Manager Salem & Hope Creek
U. S. Nuclear Regulatory Commission
One White Flint North
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11555 Rockville Pike
Rockville, MD 20852

USNRC Senior Resident Inspector - HC (X24)

Mr. K. Tosch, Manager IV
Bureau of Nuclear Engineering
PO Box 415
Trenton, NJ 08625

Document Control Desk
USNRC
Washington DC 20555

Attachment 1

Major work completed during RF12

- Completed replacement of 69 CRD Mechanism and 8 O-Ring replacements (includes Guide Tube Vacuuming of associated guide tubes). This included removal of existing CRDM Exchange Machine and replaced with new equipment
- Performed overhaul and material upgrade of the "B" and "C" Emergency Diesel Generators
- Inspections and valve replacements of "A" Service Water Loop, including coating repairs and installation of WEKO Seals
- Replaced 14 SRV Pilot Assemblies and 9 SRV Bodies
- Replaced "C" EDG Electronic Governor
- Replaced "B" EDG Electronic and Mechanical Governors
- Replaced "B" Reactor Recirc Pump Seal and implemented "B" Reactor Recirc Seal leakoff piping slope modification
- Rebuilt TACS Accumulator Floating Roof
- Repaired the "A" Control Room Chiller Labyrinth Seals
- Performed Internal inspection of all 8 MSIVs
- Repaired Drywell Insulation
- Repaired EDG Exhaust Stacks/Leaks
- Replaced 'A' & 'B' Phase Main Power Transformer
- Performed Fuel Sipping of the entire core
- Installed Digital EHC System
- Completed Reactor Level Setpoint Setdown Modification
- Implemented RX Recirc Vibration Monitoring DCP

January 11, 2005

CAL No. 1-05-001

Mr. A. Christopher Bakken, III
President and Chief Nuclear Officer
PSEG Nuclear LLC - N09
P. O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: CONFIRMATORY ACTION LETTER (1-05-001)

Dear Mr. Bakken:

On December 17, 2004, the NRC held a meeting at NRC Headquarters with Mr. M. Gallagher and other PSEG representatives to discuss questions that the NRC had regarding the vibration levels of the "B" reactor recirculation pump at Hope Creek. The vibration levels on this pump have been about twice the levels seen on the "A" recirculation pump. The vibration levels have been attributed to slight bowing of the pump shaft in the area below the seal package area which has led to frequent seal replacements. Also, industry operating experience indicates that some cracking is likely to be present in the pump shaft, which leads to questions about the expected remaining service life of the shaft. The recirculation pump forms part of the reactor coolant system boundary, and the NRC requires high reliability of that boundary during periods of plant operation.

During the December 17, 2004 meeting, you also discussed the findings of a review done for you by Sargent and Lundy (S&L) who independently assessed this vibration problem. The S&L assessment was summarized in the report, "Independent Assessment of Hope Creek Reactor Recirculation System and Pump Vibration Issues," dated November 12, 2004. The NRC had reviewed the S&L report and developed a number of questions which were provided to PSEG during December 2004, and which were discussed during the aforementioned December 17, 2004, public meeting and several subsequent teleconferences. You also addressed those issues in subsequent submittals sent to the NRC on December 29, 2004, January 4, 2005, and January 7, 2005.

The S&L Report had concluded that there is no immediate need to replace the "B" pump rotor during the current refueling outage, and the pump could be returned to service for the next operating cycle given the current level of reactor recirculation pump and piping vibrations. However, S&L recommended that both pumps be closely monitored for vibrations.

During various telephone conversations with the NRC staff in late December 2004, and early January 2005, your staff committed to a number of actions that would be taken to ensure acceptable operation of the "B" recirculation pump. These commitments are described in a subsequent letter you sent to the NRC on January 9, 2005. As a result of another telephone conversation that I had with you on January 10, 2005, it is our understanding that you have taken (or will take) all of the actions set forth in your January 9, 2005 letter, consistent with the schedule set forth therein. These commitments included: implementing a vibration-monitoring program to continuously monitor the "B" reactor recirculation pump's primary and secondary harmonic parameters (total amplitude, 1X and 2X amplitude, and 1X and 2X phase angle) during future periods of plant operation; establishing objective criteria that demonstrate that monitored parameters are within an acceptable range; developing procedures which specify the actions to be taken if the monitored parameters are outside of the specified range of the acceptance criteria [the procedures HC.OP-AB.RPV-0003(Q), HC.OP-AR.ZZ-0008(Q) Attachment E-4, and HC.ER-AP.BB-0001(Z) Rev. 0 were provided by PSEG in a letter dated January 4, 2005]; continuing this program until an inspection of the "B" reactor recirculation pump's rotating assembly and replacement of the pump shaft have been completed; notifying the NRC prior to implementing any change to this vibration monitoring and operating procedures cited above, to provide sufficient time for the NRC to complete a review of the proposed changes; and replacing the "B" reactor recirculation pump shaft and inspecting the pump's rotating assembly and pressure boundary components (such as the pump casing, etc.) at the earlier of the next refueling outage (RFO13) or during an outage of sufficient duration to accomplish pump replacement.

Pursuant to Section 182 of the Atomic Energy Act, 42 U.S.C. 2232, you are required to:

1) Notify me immediately if your understanding differs from that set forth above;

2) Notify me if for any reason you cannot complete the actions within the specified schedule and advise me in writing of your modified schedule in advance of any change; and

3) Notify me in writing when you have completed all of the actions addressed in this Confirmatory Action Letter.

Issuance of this Confirmatory Action Letter does not preclude issuance of an order formalizing the above commitments or requiring other actions on the part of the licensee; nor does it preclude the NRC from taking enforcement action for violations of NRC requirements that may have prompted the issuance of this letter. In addition, failure to take the actions addressed in this Confirmatory Action Letter may result in enforcement action.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, your submittals referenced herein (including your January 9, 2005, letter), and your response, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction. If personal privacy or proprietary Information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Sincerely,

/RA/

Samuel J. Collins
Regional Administrator

Docket No. 50-354
License No. NPF-57

cc w/encl:
M. Brothers, Vice President - Site Operations
J. T. Carlin, Vice President - Nuclear Assessment
M. Gallagher, Vice President - Engineering and Technical Support
W. F. Sperry, Director - Business Support
C. Perino, Director - Nuclear Safety and Licensing
J. A. Hutton, Hope Creek Plant Manager
R. Kankus, Joint Owner Affairs
J. J. Keenan, Esquire
M. Wetterhahn, Esquire
Consumer Advocate, Office of Consumer Advocate
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J. Lipoti Ph.D., Assistant Director of Radiation Programs, State of New Jersey
K. Tosch - Chief, Bureau of Nuclear Engineering, NJ Dept. of Environmental Protection
H. Otto, Ph.D., DNREC Division of Water Resources, State of Delaware
N. Cohen, Coordinator - Unplug Salem Campaign
W. Costanzo, Technical Advisor - Jersey Shore Nuclear Watch
E. Zobian, Coordinator - Jersey Shore Anti Nuclear Alliance

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Category
1

NRC PUBLIC MEETING FEEDBACK

Meeting Date: 01/12/2005

Meeting Title: Management meeting with PSEG to discuss the results of the NRC's special inspection and other technical issues.

In order to better serve the public, we need to hear from the meeting participants. Please take a few minutes to fill out this feedback form and return it to NRC.

1. How did you hear about this meeting?

☐ NRC Web Page ☐ NRC Mailing List ☐ Newspaper

☐ Radio/TV ☐ Other ____________

		Yes	No	Somewhat (Please explain below)
2.	Were you able to find supporting information prior to the meeting?	☐	☐	☐
3.	Did the meeting achieve its stated purpose?	☐	☐	☐
4.	Has this meeting helped you with your understanding of the topic?	☐	☐	☐
5.	Were the meeting starting time, duration, and location reasonably convenient?	☐	☐	☐
6.	Were you given sufficient opportunity to ask questions or express your views?	☐	☐	☐
7.	Are you satisfied overall with the NRC staff who participated in the meeting?	☐	☐	☐

COMMENTS OR SUGGESTIONS:

Thank you for answering these questions.

Continue Comments on the reverse.

OPTIONAL

Name ____________ Organization ____________

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☐ *Check here if you would like a member of NRC staff to contact you.*

OMB NO 3150-0197

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